

Simulation of radiation dose distribution and thermal analysis for the bulk shielding of an optimized molten salt reactor*

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The Chinese Academy of Science has launched a thorium-based molten-salt reactor (TMSR) research project with a mission to research and develop a fission energy system of the fourth generation. The TMSR project intends to construct a liquid fuel molten-salt reactor (TMSR-LF), which uses fluoride salt as both the fuel and coolant, and a solid fuel molten-salt reactor (TMSR-SF), which uses fluoride salt as coolant and TRISO fuel. An optimized 2 MWth TMSR-LF has been designed to solve major technological challenges in the Th-U fuel cycle. Preliminary conceptual shielding design has also been performed to develop bulk shielding. In this study, the radiation dose and temperature distribution of the shielding bulk due to the core were simulated and analyzed by performing Monte Carlo simulations and computational fluid dynamics (CFD) analysis. The MCNP calculated dose rate and neutron and gamma spectra indicate that the total dose rate due to the core at the external surface of the concrete wall was 1.91 $\mu\text{Sv/h}$ in the radial direction, 1.16 $\mu\text{Sv/h}$ above and 1.33 $\mu\text{Sv/h}$ below the bulk shielding. All the radiation dose rates due to the core were below the design criteria. Thermal analysis results show that the temperature at the outermost surface of the bulk shielding was 333.86 K, which was below the required limit value. The results indicate that the designed bulk shielding satisfies the radiation shielding requirements for the 2 MWth TMSR-LF.

Keywords: Molten salt reactor, High temperature, Monte Carlo calculation, Radiation dose, Neutron and gamma spectra, Thermal analysis

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I. INTRODUCTION

In 2011, the Chinese Academy of Sciences (CAS) launched a thorium-based molten-salt reactor (TMSR) related research project with a mission to research and develop a new fission energy system by 2020 [1]. The Molten Salt Reactor (MSR) [2] is a class of nuclear fission reactors using molten salt as the fission and fissile material, as well as the coolant [3]. Unlike solid-fuel reactors, the fuel is dissolved in a fluoride salt and pumped into the reactor core and the heat is absorbed and removed by the molten salt itself in MSRs [4]. MSR was selected as one of the candidates for the future development of nuclear energy by the Generation IV Forum (GIF) in 2002 [5] for its diverse advantages, such as good neutron economy, inherent safety, online reprocessing, less radioactive waste and nuclear nonproliferation [6, 7].

Since 2005, the GIF has focused on MSR from thermal spectrum molten salt reactor. Many works have been conducted to investigate the radiation safety of the MSR. Elsheikh [8] presented the features of different safety characteristics of MSR power plant and assessment in comparison to other solid fueled LWRs. Merk and Konheiser [9] applied the unstructured mesh neutron transport code HELIOS to determine the real neutron flux and power distribution for an advanced

molten salt fast reactor design. Zhu *et al.* [10] developed a one-dimensional neutron diffusion and transient analysis code to simulate neutron diffusion and transient in an MSR. In all the studies, the effect of the operation temperature was not taken into account. The TMSR project intends to construct two types of MSRs: one is a liquid fuel molten-salt reactor (TMSR-LF) which uses fluoride salt as both the fuel and coolant; the other one is a solid fuel molten-salt reactor (TMSR-SF), which uses fluoride salt as a coolant and TRISO fuel. A preliminary conceptual design has been carried out to develop a 2 MWth liquid fuel molten-salt reactor. Primary shielding calculations and design have also been performed for the 2 MWth TMSR-LF. The primary goal of radiation shielding is to protect personnel from radiation hazards. The minimum thickness of materials to decrease the dose rate to an acceptable level should be determined. Another key point is the high operation temperature of the molten salt fuel. The fuel was designed to enter the core of the 2 MWth TMSR-LF at 908.15 K. The high temperature around the core was not suitable for shielding materials. So a water-cooled, cooling fins-filled thermal shield was designed to remove heat and bring down the temperature. The performance of the designed bulk shielding at attenuating the dose rate and bringing down the temperature should be assessed to confirm the radiation shielding safety for the 2 MWth TMSR-LF. It is important to note that only the core of the 2 MWth TMSR-LF was considered as the radiation source, the auxiliary components such as the pipes of the primary system, the heat exchanger (IHx), and the pump were not taken into account in the analysis.

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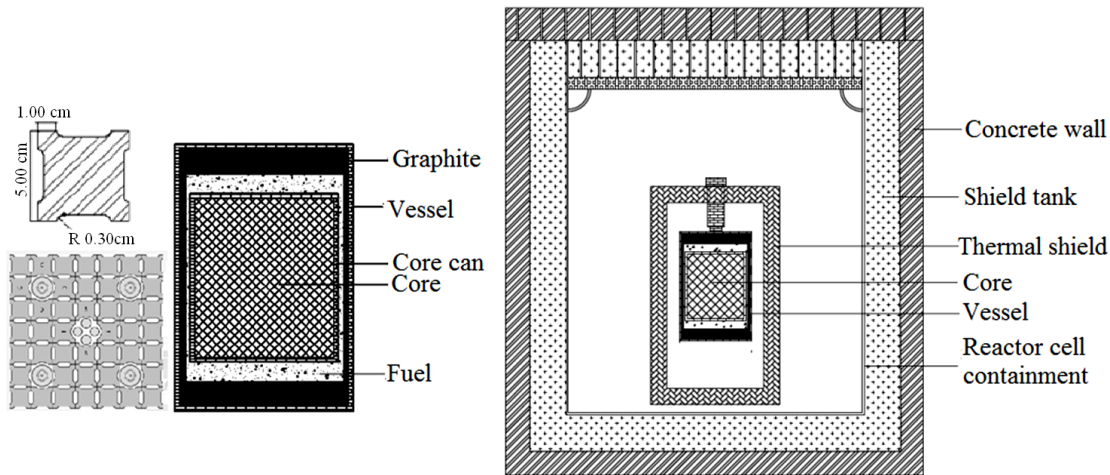


Fig. 1. Configuration of the core and the bulk shielding of the 2 MWth TMSR-LF.

TABLE 1. Parameters about the fuel and the structures of the designed bulk shielding

	General	Structures	ID (cm)	OD (cm)	Height (cm)	Materials
Fuel compositions	BeF ₂ -LiF-ZrF ₄ -UF ₄ 65-29.2-5-0.8 mole%	Core	-	140	160	-
Uranium fractions	²³⁴ U- ²³⁵ U- ²³⁶ U- ²³⁸ U 0.3-35-0.3-64.4 atom%	Core can	140	146	166	Hastelloy-N
Density	2270 kg/m ³	Fuel	146	156	206	
Mass flow rate	171.86 kg/s	Reflector	156	166	256	Graphite
Fuel inlet temperature	908.15 K	Vessel	166	192	262	Hastelloy-N
Fuel outlet temperature	936 K	Thermal shield	240	320	540	Stainless steel & water
		Reactor cell containment	730	740	670	Stainless steel
Fuel viscosity	$\mu[\text{Pas}] = 0.1542e^{3624/T_t}$	Shield tank	740	900	835	Magnetite concrete
Fuel heat capacity	1976 J/(kg K)	Concrete wall	900	1060	995	Ordinary concrete
Fuel thermal conductivity	1.44 W/(m K)					
Graphite heat capacity	1758 J/(m K)					
Graphite thermal conductivity	$\lambda[\text{W}/(\text{m} \cdot \text{K})] = 3763 \times T_g^{-0.7}$					

The As Low As Reasonably Achievable (ALARA) principle was used as guidance for radiation protection in the 2 MWth TMSR-LF. Furthermore, the national standard for radiation source safety [11] and the associated guidance were utilized in the shielding design and analysis. Taking into account these guidelines, shielding analyses were performed for the entire bulk shielding. Simulation of the radiation dose distribution and the neutron spectra due to the core were carried out for the bulk shielding of the 2 MWth TMSR-LF by using the Monte Carlo method. Thermal analysis was also carried out to present the temperature distribution of the designed bulk shielding for this high temperature reactor.

II. METHOD

A. Description of the 2 MWth TMSR-LF and bulk shielding

The 2 MWth TMSR-LF considered in this work is a 2 MWth graphite-moderated thermal reactor making use of

solutions of uranium in fluoride carrier salts. A previous parametric study was performed to develop this optimized MSR based on the Molten-Salt Reactor Experiment (MSRE) [12], which was originally developed at Oak Ridge National Laboratory (ORNL) in the 1960s.

Shielding calculations were carried out and shielding design was presented for the 2 MWth TMSR-LF previously. The designed bulk shielding consists of the thermal shield, the shield tank, and the concrete wall, as shown in Fig. 1. Table 1 gives information about the 2 MWth TMSR-LF and the bulk shielding. The reactor core, which was located in the center of the bulk shielding, was a 140 cm (diameter) by 160 cm (high) graphite matrix structure made up of 513 graphite stringers. The graphite stringers were mounted in a vertical close-packed array, as shown in Fig. 1. Half-channels were machined in the four faces of each stringer to form flow passages in the assembly. In addition, the core of the optimized MSR was surrounded by a graphite lump, which not only serves as the moderator, but also provides a neutron reflector to gain good neutron economics. The optimized MSR was designed to be fueled with BeF₂-LiF-ZrF₄-UF₄ (65-29.2-5-0.8 mol%),

with atomic fractions of uranium of 0.3% ^{234}U , 35% ^{235}U , 64.4% ^{238}U and 0.3% ^{236}U .

The thermal shield was a water-cooled, cooling fins-filled stainless steel container completely surrounding the reactor core. The thermal shield was designed to reduce the neutron dose rate by a factor of 10^4 and attenuate the gamma dose rate by a factor of about 10^3 in the horizontal plane. As mentioned above, the thermal shield was required to absorb the heat from the core significantly. It was designed so that the difference in temperature at the internal and external surface of the thermal shield was about 150 K.

The fuel pump and heat exchanger were located in a 370 cm diameter by 670 cm high reactor cell. This reactor cell was surrounded by a magnetite concrete shield tank, which in turn sits within an ordinary concrete wall [13]. The reactor cell was not allowed to be entered under any circumstances after the reactor was operated at power because of the high radiation level. The total dose rate criteria for the radiation shielding of the 2 MWth TMSR-LF was $2.5\ \mu\text{Sv/h}$ at areas outside the concrete wall.

Due to the principle of the temperature limit of the shielding materials, it was required by the ASME that the maximum temperature of the concrete should not exceed 366 K during operation [14].

B. Method of analysis

The Monte Carlo code MCNP5 was employed as a calculation tool for numerical radiation transport studies. MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. For the shielding calculation of the 2 MWth TMSR-LF, the fuel was considered to be static in the core. The total neutrons from the core and the fission gamma rays were considered as the radiation source in shielding analysis. The fission gamma rays were determined by source term calculation according to the operation plan. A MCNP model of the core and its bulk shielding was developed following the geometry structure and dimensions given in Table 1 for the Monte Carlo simulations. The neutron and gamma spectra averaged over the core were calculated, then the spectra of each shield were simulated and converted to the dose rate by employing the flux-to-dose rate conversion factors recommended by ICRP Publication 74 [15]. The data libraries used in this calculation were processed by NJOY, based on the most recent ENDF/B-VII nuclear data available.

The computational fluid dynamics (CFD) code FLUENT was applied to the thermal analysis for the 2 MWth TMSR-LF and its bulk shielding. 3D model of the core, and shields were set up by employing the 3D mechanical computer-aided design (CAD) program of SolidWorks. GAMBIT, a general-purpose preprocessor for CFD analysis, was applied to generate meshes and define the boundary types of the model. The meshed model was then imported to FLUNET to simulate the temperature field of the 2 MWth TMSR-LF and the bulk shielding. Materials, such as water and steel, were con-

tained in the FLUENT database, but other materials, such as the fuel, the graphite, and the Hastelloy-N [16], still need to be defined by the user. Boundary conditions, such as the flow rate of the fuel and the fuel temperatures at the inlet and outlet also need to be defined. The parameters for the CFD simulation are listed in Table 1. The heat source was defined based on the MCNP5 calculated power density of the core through user defined functions (UDF) [17–19].

III. RESULTS AND DISCUSSION

A. Radiation dose distribution

The neutron and gamma spectra, as well as the power density, averaged over the core were analyzed first. The output of the MCNP5 was defined by a number of tally specification cards (Fn card). The Monte Carlo program provides seven basic neutron tally types and six standard photon tallies. For the measurements of the neutron and gamma spectra, the tally for flux averaged over a cell (tally F4) was used. For the measurements of power density, the tally for fission energy deposition averaged over a cell (tally F7) was used. And the KCODE was employed to specify the source of the 2 MWth TMSR-LF. 15 550 cycles with 10^6 particles per cycle were run for the simulation, and 50 cycles were skipped before the tally accumulation began.

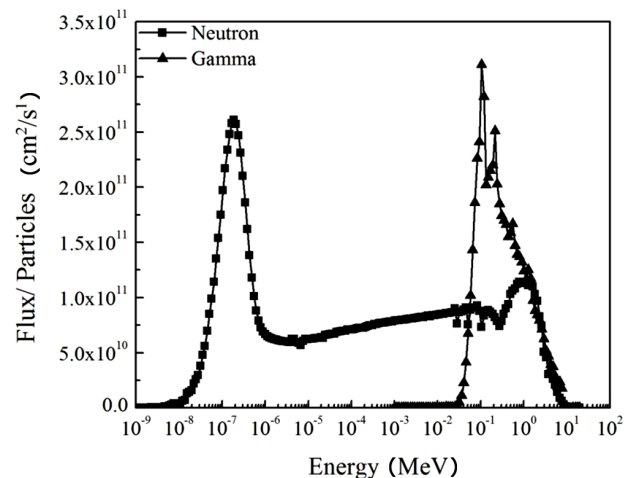


Fig. 2. Neutron and gamma spectra averaged over the core of the 2 MWth TMSR-LF.

The neutron and gamma spectra averaged over the core were analyzed and shown in Fig. 2. Notice that the neutrons are spread widely from 10^{-9} MeV to about 20 MeV. The peak in the neutron spectrum where most neutrons appear in the 2 MWth TMSR-LF was between 0.01 eV and 1 eV. The gamma rays have a relatively narrow bandwidth where the energy is spread from 1 keV to about 20 MeV. There were two sharp peaks in the gamma spectrum around 0.1 MeV.

Figure 3 indicates that the power distribution was symmetric in the axial direction of the core and the peak appears at the middle of the axis. The peak of the power did not appear

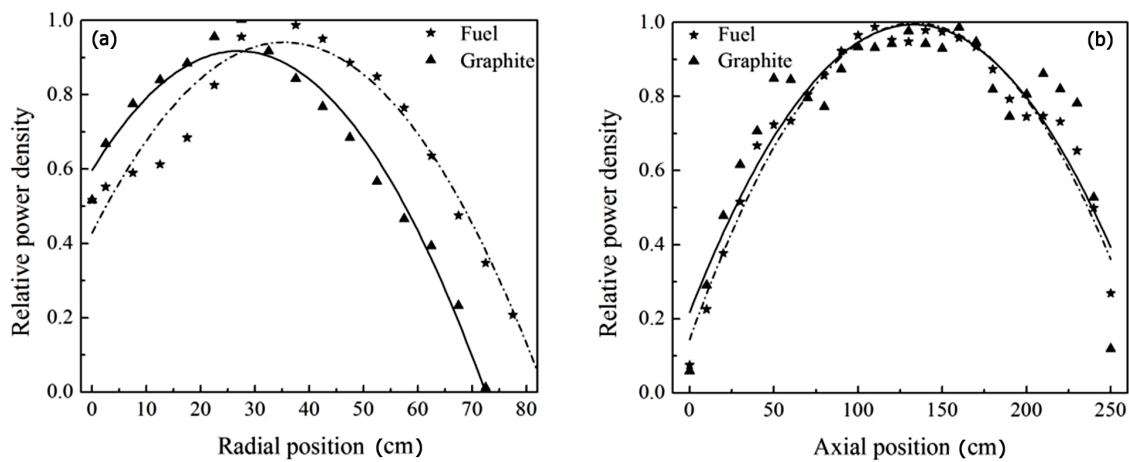


Fig. 3. Power density of the 2 MWth TMSR-LF.

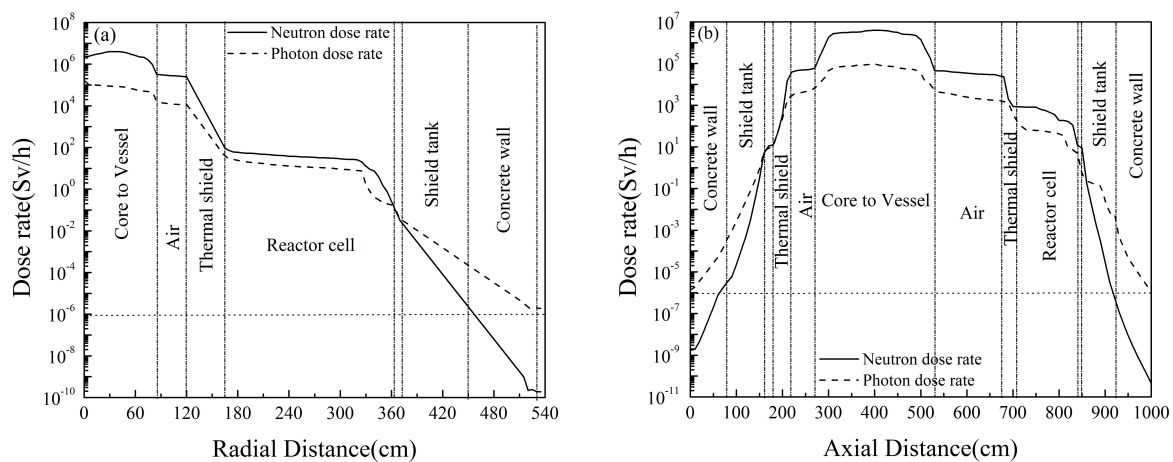


Fig. 4. Radiation dose rate distributions of the 2 MWth TMSR-LF in the (a) radial and (b) axial direction.

at the center of the radial direction because there were control rods at that position, as seen in Fig. 1.

A transport calculation was carried out to simulate the radiation dose distribution under the condition of the maximum power in Fig. 3 to obtain a conservative estimate. For the measurements of the neutron and gamma dose rate, the flux-to-dose rate conversion factors written in the Dose Energy Card (DE) and Dose Function Card (DF) were used. Figure 4(a) presents the MCNP5 calculation results of the radiation dose rate at the radial direction along the center line of the core. Notice that the magnetite concrete thickness of 80 cm was sufficient to reduce the neutron dose rate to $1 \mu\text{Sv/h}$. However, an additional 80 cm of ordinary concrete may be required to attenuate the photon dose rate. The total dose rates due to the core at the external surface of the concrete wall were $1.91 \mu\text{Sv/h}$. Figure 4(b) shows the radiation dose rate at the axial direction as about 40 cm from the center line of the core, at which position the peak of the power appears. The total dose rate due to the core was $1.16 \mu\text{Sv/h}$ above and $1.33 \mu\text{Sv/h}$ below the bulk shielding.

It also can be noticed from Fig. 4 that the neutron dose rate due to the core was at the 10^5 level at the internal surface and the 10^2 level at the external surface of the thermal shield. The photon dose rate due to the core was at the 10^4 level at the internal surface of the thermal shield and at the 10 level at the external surface. The attenuate factors of the neutron dose rate and photon dose rate meet the design objective.

Figure 5 shows the neutron and gamma spectra outside of each shield at the radial direction along the center line of the core. The spectra indicate that the thermal shield has extraordinary attenuation in neutron flux and gamma flux.

B. Temperature distribution

As mentioned above, the molten salt was designed to be pumped into the core at 908.15 K, with a flow rate of 171.86 kg/s. Thermal analysis needs to be carried out to present the temperature distribution of the complex. The thermal analysis was displayed using the CFD code FLUENT.

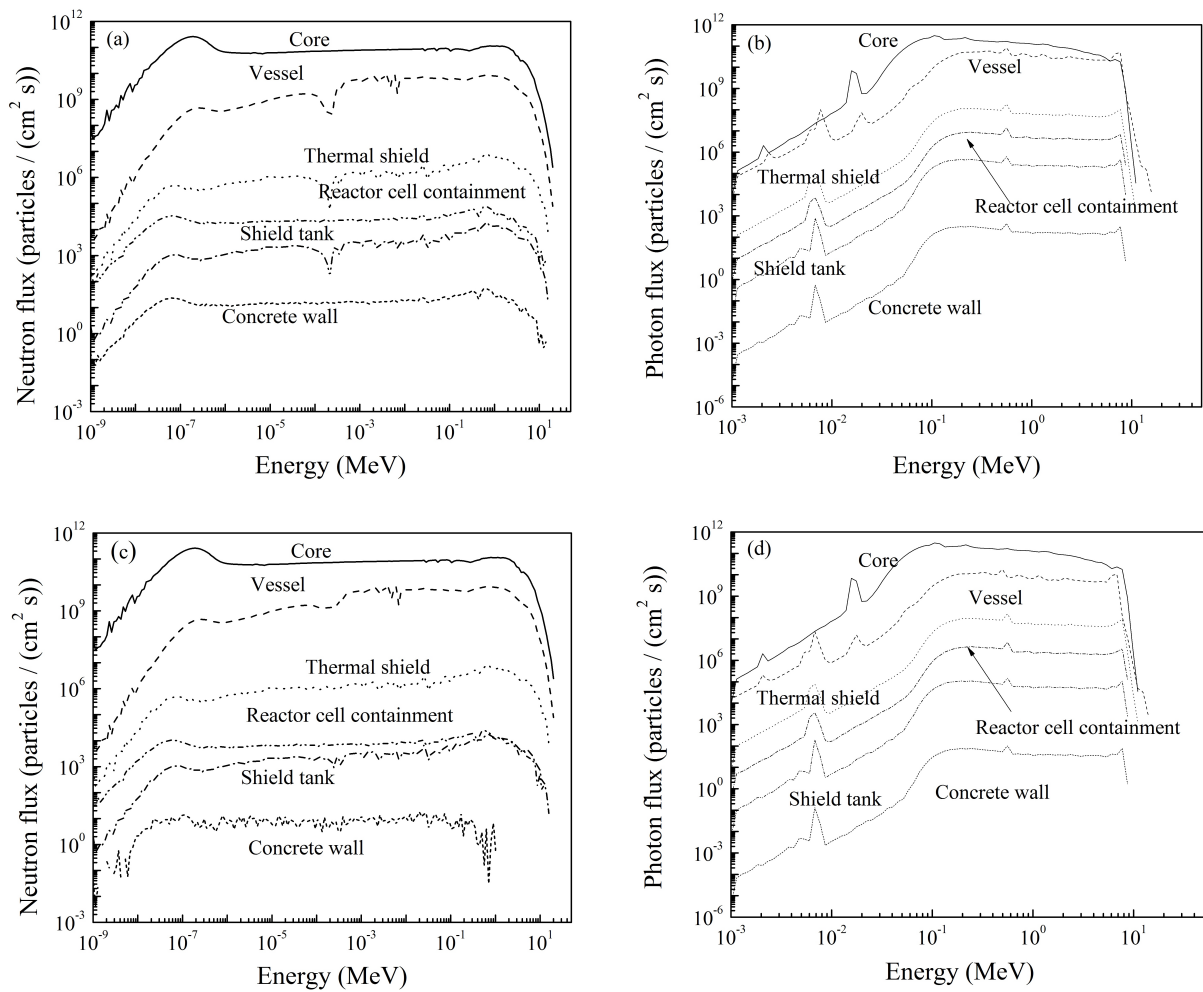


Fig. 5. Spectra of each shield (a) neutron in the radial direction; (b) gamma in the radial direction; (c) neutron in the axial direction; (d) gamma in the axial direction.

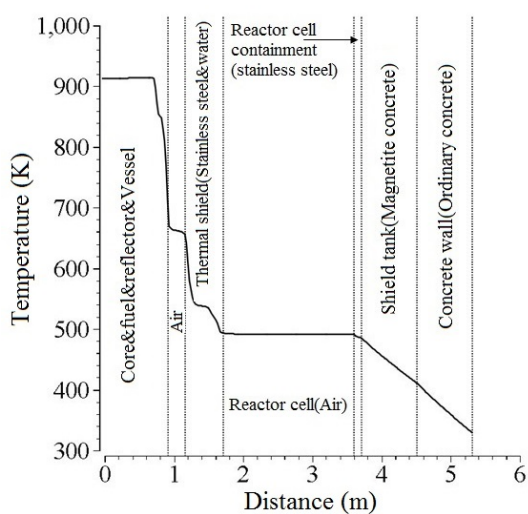


Fig. 6. Temperature profile of the bulk shielding.

The porous model [20, 21] was employed to simulate the core because the dimensions of the fuel passages were relatively small compared to the dimensions of the core. The power destiny distribution shown in Fig. 3 was loaded into the CFD model as the heat source through the UDF compiled in the C programming language.

Figure 6 gives the temperature profile of the bulk shielding at the radial direction along the center line of the core, with the flow rate of the cooling water in the thermal shield at 6.3 kg/s. Notice that the temperature difference between the internal surface and external surface of the thermal shield was about 150 K, which meets the design objective. The temperature at the internal surface of the concrete wall was 333.86 K, which was below the required limit value.

IV. CONCLUSION

The radiation dose distribution and temperature distribution of the 2 MWth TMSR-LF for the proposed TMSR project

was simulated and analyzed by performing Monte Carlo simulations and CFD analysis. The neutron and gamma spectra from each shield were first calculated by MCNP5, then calculations were performed to present the total radiation dose distribution, based on these spectra, by using the flux-to-dose rate conversion factors. The calculated results indicate that the total dose rate due to the core at the external surface of the concrete wall was 1.91 $\mu\text{Sv/h}$ in the radial direction and 1.16 $\mu\text{Sv/h}$ above and 1.33 $\mu\text{Sv/h}$ below the bulk shielding. The neutron and gamma spectra show that the thermal shield, composed of stainless steel fins and cooling water, attenuates

the neutron and gamma flux extraordinarily.

Thermal analysis was also carried out by employing FLUENT. The porous model was employed to simulate the core and the power density calculated by MCNP5 was loaded to the CFD model as the heat source through the UDF. Results show that the thermal shield could reduce the temperature sharply and the temperature of the concrete was below the design objective.

Results of the present analysis show that the designed bulk shielding satisfies the shielding requirements of the 2 MWth TMSR-LF.

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- [1] Jiang M H, Xu H J and Dai Z M. Advanced fission energy program-TMSR nuclear energy system. *Bull Chin Acad Sci*, 2012, **03**: 366–374. DOI: [10.3969/j.issn.1000-3045.2012.03.016](https://doi.org/10.3969/j.issn.1000-3045.2012.03.016)
 - [2] MacPherson H G. The molten salt reactors adventure. *Nucl Sci Eng*, 1985, **90**: 374–380.
 - [3] Merle L E, Heuer D, Allibert M, *et al.* Introduction to the physics of molten salt reactors. Corsica (France): Springer Press, 2008, 501–521.
 - [4] Ergen W K, Callihan A D and Mills C B. The aircraft reactor experiment-physics. *Nucl Sci Eng*, 1957, **2**: 826–840.
 - [5] DOE. A technology roadmap for Generation IV nuclear energy systems. Nuclear Energy Research Advisory Committee and the Generation IV International Forum, GIF-002-00, 2002.
 - [6] Nuttin A, Heuer D, Billebaud A, *et al.* Potential of thorium molten salt reactors: Detailed calculations and concept evolutions in view of a large nuclear energy production. *Prog Nucl Energ*, 2005, **46**: 77–99. DOI: [10.1016/j.pnucene.2004.11.001](https://doi.org/10.1016/j.pnucene.2004.11.001)
 - [7] Moir R W and Teller E. Thorium-fueled underground power plant based on molten salt technology. *Nucl Technol*, 2005, **151**: 334–340.
 - [8] Elsheikh B M. Safety assessment of molten salt reactors in comparison with light water reactors. *J Radiat Res Appl Sci*, 2013, **6**: 63–70. DOI: [10.1016/j.jrras.2013.10.008](https://doi.org/10.1016/j.jrras.2013.10.008)
 - [9] Merk B and Konheiser J. Neutron shielding studies on an advanced molten salt fast reactor design. *Ann Nucl Energy*, 2014, **64**: 441–448. DOI: [10.1016/j.anucene.2013.07.014](https://doi.org/10.1016/j.anucene.2013.07.014)
 - [10] Zhu L, Pu P, Du S, *et al.* Simulation of neutron diffusion and transient analysis of MSR. *Nucl Sci Tech*, 2014, **25**: 020601. DOI: [10.13538/j.1001-8042/nst.25.020601](https://doi.org/10.13538/j.1001-8042/nst.25.020601)
 - [11] Basic standards for protection against ionizing radiation and for the safety of radiation sources. GB18871, 2002.
 - [12] Robertson R C. MSRE design and operations report part I: Description of reactor design. ORNL USA, ORNL-TM-0728, 1965.
 - [13] McConn R J, Gesh C J, Pagh R T, *et al.* PIET-43741-TM-963, PNNL-15870 Rev. 1, Compendium of material composition data for radiation transport modeling. USA: Pacific Northwest National Laboratory, 2011.
 - [14] ASME. BPVC, Section III-Rules for construction of nuclear facility components. Division 2-Code for concrete containments. ASME, 2013.
 - [15] ICRP. Conversion coefficients for use in radiological protection against external radiation. ICRP Publication 74, 1996.
 - [16] Haynes International I. HASTELLOY®N alloy, Indiana USA, 2002.
 - [17] He W F, Dai Y P, Wang J F, *et al.* Performance prediction of an air-cooled steam condenser using UDF method. *Appl Therm Eng*, 2013, **50**: 1339–1350. DOI: [10.1016/j.applthermaleng.2012.06.020](https://doi.org/10.1016/j.applthermaleng.2012.06.020)
 - [18] Liu Y, Li W Z and Quan S L. A self-standing two-fluid CFD model for vertical upward two-phase annular flow. *Nucl Eng Des*, 2011, **241**: 1636–1642. DOI: [10.1016/j.nucengdes.2011.01.037](https://doi.org/10.1016/j.nucengdes.2011.01.037)
 - [19] Ye F, Xiao J, Hu B, *et al.* Implementation for model of adsorptive hydrogen storage using UDF in fluent. *Phys Procedia*, 2012, **24**: 793–800. DOI: [10.1016/j.phpro.2012.02.118](https://doi.org/10.1016/j.phpro.2012.02.118)
 - [20] Pilehvar A F, Aghaie M, Esteki M H, *et al.* Evaluation of compressible flow in spherical fueled reactors using the porous media model. *Ann Nucl Energy*, 2013, **57**: 185–194. DOI: [10.1016/j.anucene.2013.01.062](https://doi.org/10.1016/j.anucene.2013.01.062)
 - [21] Short M P, Hussey D, Kendrick B K, *et al.* Multiphysics modeling of porous CRUD deposits in nuclear reactors. *J Nucl Mater*, 2013, **443**: 579–587. DOI: [10.1016/j.jnucmat.2013.08.014](https://doi.org/10.1016/j.jnucmat.2013.08.014)